ACCESSION #: 9604160090

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Donald C. Cook Nuclear Plant - Unit 1 PAGE: 1 OF 3

DOCKET NUMBER: 05000315

TITLE: Reactor Trip on Low Feedwater Flow Coincident with Low

Steam Generator Level Due to Failure of the Feedwater

Differential Pressure Controller

EVENT DATE: 03/17/96 LER #: 96-002-00 REPORT DATE: 04-10-96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Paul Schoepf, Plant Engineering TELEPHONE: (616) 465-5901,

Superintendent x2408

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 1527 hours on March 17, 1996, with Unit 1 at 100% Rated Thermal Power, a reactor trip signal was received. This signal was generated as a result of low feedwater flow coincident with low Steam Generator level in the number 4 Steam Generator. The low level resulted from the failure of the Main Feedpump (MFP) differential pressure controller, which automatically controls the differential pressure between feedflow

pressure into the SG, and main steam pressure exiting the Steam Generator. The Operators took manual control of the feedpumps to regain level in the Steam Generator, however, the reactor trip logic conditions were met before level could be restored and the trip signal was generated.

All safety systems operated normally in response to the trip signal and the unit was stabilized in Mode 3. The event was determined to have been caused by a failure in an integrated circuit logic chip in the main feedpump differential pressure controller. The failed controller was replaced and subsequently returned to the manufacturer for further analysis. The event was determined to have no actual or potentially adverse effect on the health and safety of the public.

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Conditions Prior to Occurrence:

Unit 1 in Mode 1, Power Operations, at 100 percent Rated Thermal Power

Description of Event:

At 1527 hours on March 17, 1996, Unit 1 tripped from 100% Rated Thermal Power as a result of low feedwater flow coincident with low Steam Generator (SG) level in the number 4 SG. The low level resulted from the failure of the Main Feedpump (MFP) differential pressure controller (EIIS/JB-PDC), which automatically controls the differential pressure between feedflow pressure into the SG and Main Steam pressure exiting the SG.

The controller failure caused the speed, and thus discharge pressure, of both main feedpumps to decrease. Hence the main feedwater flow to all 4 Steam Generators decreased below that necessary to maintain Steam Generator level, and when two-out-of-three level instruments in the number 4 Steam Generator reached the low level setpoint, the units reactor automatically tripped. The operators took manual control of each of the feedpump controllers to restore the main feedwater pumps' speed in an effort to increase the feedwater flow to the Steam Generators before the unit tripped. However, the logic conditions were met before level could be restored and the trip signal was generated.

All safety systems operated normally in response to the trip signal. Originally, it had been suspected that a Main Steam Safety Valve had lifted, but review of the separate Steam Generator pressure traces revealed no characteristics normally attributed to the opening and resealing of a safety relief valve. It has been concluded that an Auxiliary Steam safety associated with the condensate system lifted. The four hour ENS phone call stated that the Motor Driven Auxiliary Feedpumps had been manually started by the operators, however, it was later determined that the pumps had automatically started.

Post-trip temperature decreased to approximately 535 degree Fahrenheit due to leak-by of two secondary system valves from the main steam lines to the moisture separator reheaters. Manual isolation and tightening of these valves by the operators arrested the temperature decrease and restored T sub avg to the 547 degree Fahrenheit value. The unit was stabilized in Mode 3, and the differential pressure controller was replaced.

Cause of the Event:

This event was caused by a failure in the main feedpump differential pressure controller. The controller that failed has been returned to the manufacturer for failure determination. There were no other obvious indications of failure on the controller circuit cards such as charred components or appearances of poor connections.

Analysis of Event:

This event is being reported per 10 CFR 50.73(a)(2)(iv) as an event that resulted in automatic actuation of Engineered Safety Features (ESF), including the Reactor Protection System (RPS).

A reactor trip due to low main feedwater flow in coincidence with low Steam Generator level in the number 4 Steam Generator. All control rods inserted, both Motor Driven Auxiliary Feedwater Pumps and the Turbine Driven Auxiliary Feedwater automatically started, and a Feedwater Isolation occurred.

The unit successfully transferred to the normal offsite power supply, and the emergency diesel generators remained in standby. This event did not have any actual or potentially adverse impact on the health and safety of the public.

Corrective Action:

The failed feedpump differential pressure controller has been replaced. The failed controller has been returned to the manufacturer for further analysis. Discussions have been initiated with the controller manufacturer to determine the failure mechanism and explore actions to mitigate the consequences of future failures.

Failed Component Identification:

Component Name: 1-RU-5, Feed pump Differential Pressure Controller

Manufacturer: Kent-Taylor

Model: XL-170

EIIS Code: JB-PDC

Previous Similar Events:

None

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American Electric Power

Cook Nuclear Plant

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Bridgman, MI 49106

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AMERICAN

ELECTRIC

POWER

April 10, 1996

United States Nuclear Regulatory Commission

Document Control Desk

Rockville, Maryland 20852

Operating Licenses DPR-58

Docket No. 50-315

Document Control Manager:

In accordance with the criteria established by 10 CFR 50.73 entitled

Licensee Event Report System, the following report is being submitted:

96-002-00

Sincerely,

A. A. Blind

Site Vice President

/clc

Attachment

c: H. J. Miller, Region III

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